Dominion Energy Kewaunee, Inc.
5000 Dominion Boulevard, Glen Allen, VA 23060

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U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Serial No. 10-047
LIC/JG/R0
Docket No.: 50-305
License No.: DPR-43

DOMINION ENERGY KEWAUNEE, INC.
KEWAUNEE POWER STATION
LICENSE AMENDMENT REQUEST 246: REVISION TO REACTOR COOLANT SYSTEM PRESSURE AND TEMPERATURE LIMITS AND LOW TEMPERATURE OVERPRESSURE PROTECTION LIMITATIONS

Pursuant to 10 CFR 50.90, Dominion Energy Kewaunee, Inc. (DEK) requests an amendment to Facility Operating License Number DPR-43 for Kewaunee Power Station (KPS). This proposed amendment would revise the Operating License by modifying KPS Technical Specification (TS) 3.1.a.1.C, "Reactor Coolant Pumps;" TS 3.1.a.3, "Pressurized Safety Valves," and TS 3.1.b, "Heatup and Normal Cooldown Limit Curves for Normal Operation." The revised requirements would consist of new heatup and cooldown pressure-temperature (P/T) limit curves and a higher LTOP enabling temperature. The revisions would extend the applicability of the P/T limit curves and LTOP enabling temperature through the expected period of a KPS extended license.

DEK plans to initiate use of the proposed P/T and LTOP limitations during startup of fuel cycle 31 (Spring 2011). Therefore, DEK requests approval of the proposed amendment by February 28, 2011. Once approved, the amendment shall be implemented within 120 days.

Attachment 1 to this letter contains a description, analysis, significant hazards determination, and environmental considerations for the proposed changes. Attachment 2 contains the marked-up Technical Specifications. Attachment 3 contains marked-up Technical Specifications Bases pages. Also, enclosed are five Westinghouse Electric Company reports that provide technical justification for this proposed amendment.

The KPS Facility Safety Review Committee has approved the proposed change and a copy of this submittal has been provided to the State of Wisconsin in accordance with 10 CFR 50.91(b).

A001
NRR

If you have questions or require additional information, please contact Mr. Jack Gadzala at 920-388-8604.

Very truly yours,

J. Alan Price
Vice President – Nuclear Engineering

COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by J. Alan Price, who is Vice President Nuclear Engineering of Dominion Energy Kewaunee, Inc. He has affirmed before me that he is duly authorized to execute and file the foregoing document in behalf of that Company, and the statements in the document are true to the best of her knowledge and belief.

Acknowledged before me this 13TH day of April, 2010.

My Commission expires: MAY 31, 2010.

VICKI L. HULL
Notary Public
Commonwealth of Virginia
140542
My Commission Expires May 31, 2010

Vicki L. Hull
Notary Public

Attachments:

1. Discussion of Change, Analysis, Significant Hazards Determination, and Environmental Considerations
2. Marked-up Technical Specifications Pages
3. Marked-up Technical Specifications Bases Pages

Enclosures:

1. WCAP-15074, Revision 1, "Evaluation of the 1P3571 Weld Metal from the Surveillance Programs for Kewaunee and Maine Yankee," dated August 2006.
2. WCAP-16609-NP, Revision 0, "Master Curve Assessment of Kewaunee Power Station Reactor Vessel Weld Metal," dated October 2006.
3. WCAP-16641-NP, Revision 0, "Analysis of Capsule T from Dominion Energy Kewaunee Power Station Reactor Vessel Radiation Surveillance Program," dated October 2006.
4. WCAP-16642-NP, Revision 0, "Evaluation of Pressurized Thermal Shock for Kewaunee Power Station," dated December 2006.
5. WCAP-16643-NP, Revision 2, "Kewaunee Power Station Heatup and Cooldown Limit Curves for Normal Operation," dated February 2008.

Commitments made by this letter: None

cc: Regional Administrator, Region III
U. S. Nuclear Regulatory Commission
2443 Warrenville Road
Suite 210
Lisle, IL 60532-4352

Mr. P. S. Tam
Sr. Project Manager
U.S. Nuclear Regulatory Commission
One White Flint North, Mail Stop O8-H4A
11555 Rockville Pike
Rockville, MD 20852-2738

NRC Senior Resident Inspector
Kewaunee Power Station

Public Service Commission of Wisconsin
Electric Division
P.O. Box 7854
Madison, WI 53707

ATTACHMENT 1

LICENSE AMENDMENT 246
REACTOR COOLANT SYSTEM PRESSURE AND TEMPERATURE LIMITS AND LOW TEMPERATURE OVERPRESSURE PROTECTION

DISCUSSION OF CHANGE, ANALYSIS, SIGNIFICANT HAZARDS DETERMINATION, AND ENVIRONMENTAL CONSIDERATIONS

KEWAUNEE POWER STATION
DOMINION ENERGY KEWAUNEE, INC.

LICENSE AMENDMENT 246
REVISION TO REACTOR COOLANT SYSTEM PRESSURE AND TEMPERATURE LIMITS AND LOW TEMPERATURE OVERPRESSURE PROTECTION LIMITATIONS

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1.0 DESCRIPTION

This license amendment request proposes to revise the current heatup and cooldown pressure-temperature (P/T) limit curves and low temperature overpressure protection (LTOP) requirements for the Kewaunee Power Station (KPS).

2.0 PROPOSED CHANGE

The proposed amendment would modify the KPS Technical Specifications (TS) associated with reactor coolant system (RCS) temperature and pressure. The proposed amendment would extend the applicability period of the heatup and cooldown P/T limit curves and LTOP limitations from the currently existing 31.1 effective full power years (EFPY) to the proposed value of 52.1 EFPY. The revised requirements consist of new heatup and cooldown P/T limit curves and a higher LTOP enabling temperature. The revision would extend the applicability of the P/T limit curves and LTOP enabling temperature through the expected period of a KPS extended license.

2.1 Custom Technical Specifications Changes

The scope of proposed changes under License Amendment Request (LAR) 246 applies to KPS Custom TS (CTS) Section TS 3.1, "Reactor Coolant System," and Figures TS 3.1-1 and TS 3.1-2. The proposed change will modify the limiting conditions for operation for reactor coolant system temperature and pressure as summarized below:

1. TS 3.1.b.1 is modified to address applicability of Figure TS 3.1-1 and Figure TS 3.1-2 for a service period of up to 52.1 EFPY. The present footnote in each of the figures stating that the curves are limited to 31.1 EFPY is deleted, because the proposed new pressure and temperature limitation curves are applicable through 52.1 EFPY. This change makes TS 3.1.b.1 consistent with the expiration date noted on the revised heatup and cooldown P/T limit curves.
2. The current enabling temperature for the LTOP system (specified in TS 3.1.a.1.C, TS 3.1.a.3, TS 3.1.b.1.C, and TS 3.1.b.4) of 200 °F is being changed to 343 °F. This change makes TS 3.1.a.1.C, TS 3.1.a.3, TS 3.1.b.1.C, and TS 3.1.b.4 consistent with the value of RT_{NDT} (where RT_{NDT} is the adjusted reference temperature, including margin, at the quarter thickness location) applied to the revised heatup and cooldown P/T limit curves. The 1998 Edition (through 2000 Addendum) of the American Society of Mechanical Engineers (ASME) Boiler & Pressure Vessel Code (B&PVC) Section XI, Appendix G states that:

    a) LTOP systems shall be effective at coolant temperatures less than 200 °F or at coolant temperatures corresponding to a reactor vessel metal temperature less than RT_{NDT} + 50 °F, whichever is greater.

    b) LTOP systems shall limit the maximum pressure in the vessel to 100% of the pressure determined to satisfy $2K_{Im} + K_{It} < K_{Ic}$ (Equation (1)).

The LTOP enabling temperature value of 343 °F is determined from the value of RT_{NDT} for the limiting material, which is the circumferential weld at end of license extension (EOLE) fluence (278 °F), plus 50 °F, plus 15 °F (bounding anticipated temperature difference between the coolant temperature and vessel metal temperature) (Enclosure 5).

3. Figure TS 3.1-1 and Figure TS 3.1-2 have been modified using the NRC-approved methodology to meet the requirements of 10 CFR 50, Appendix G, for pressure and temperature (P/T) limitations applicable during a service period of up to 52.1 EFPY. The new figures provided in Attachment 2 replace, in their entirety, the corresponding figures in the current KPS TS.
4. TS Basis Section 3.1 has been revised accordingly (including footnote revisions).
5. The list of TS figures has been modified to address the new applicability period of 52.1 EFPY for Figure TS 3.1-1 and Figure TS 3.1-2.

2.2 Improved Technical Specifications Changes

Dominion Energy Kewaunee (DEK) submitted a license amendment request (LAR 249) on August 24, 2009, proposing to revise the KPS current Custom TS (CTS) to Improved TS (ITS), consistent with the Improved Standard Technical Specifications (ISTS) described in NUREG 1431, "Standard Technical Specifications - Westinghouse Plants," Revision 3.0.

The amendment proposed in this request (LAR 246), will require changes to the ITS as proposed in the August 24, 2009 amendment request (LAR 249). This amendment will require modification to the proposed ITS 3.4.3, "Reactor Coolant System (RCS) Pressure and Temperature (P/T) Limits," and ITS 3.4.12, "Low Temperature Overpressure Protection (LTOP) System." The revised requirements would consist of new heatup and cooldown P/T limit curves and a higher LTOP enabling temperature (identical to the changes being proposed in this amendment). The revisions would correspondingly extend the applicability of the heatup and cooldown P/T limit curves through 52.1 EFPY.

However, the justification and basis for the associated changes to transition to the proposed ITS are independent of, and not germane to, the requested changes proposed to the CTS in this amendment request. As such, a corresponding request to revise the proposed ITS wording will be submitted separately in coordination with NRC review of the ITS amendment.

3.0 BACKGROUND

Components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. Kewaunee TS 3.1, "Reactor Coolant System," limits the pressure and temperature changes during RCS heatup and cooldown to within the design assumptions and the stress limits for cyclic operation. This TS provides P/T limit curves for heatup, cooldown, inservice leak and hydrostatic testing, criticality, and limits for the maximum rate of change of reactor coolant temperature.

10 CFR 50, Appendix G requires the establishment of P/T limits for specific material fracture toughness requirements of the reactor coolant pressure boundary (RCPB) materials. This regulation specifies an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of ASME Section III and XI, Appendix G.

The Low Temperature Overpressure Protection (LTOP) system limits RCS pressure at low temperatures so the integrity of the RCPB is not compromised by violating the pressure and temperature (P/T) limits of 10 CFR 50, Appendix G. The reactor vessel is the limiting RCPB component for demonstrating such protection. Kewaunee TS 3.1, "Reactor Coolant System," provides the maximum RCS pressure for the existing RCS cold leg temperature during cooldown, shutdown, and heatup to meet the requirements when the RCS is less than or equal to the LTOP enabling temperature.

The current heatup and cooldown limit curves and LTOP limitations are valid through 31.1 EFPY, which is projected to occur during operating cycle 31 (approximately April 2012). At the time the existing heatup and cooldown limit curves and LTOP limitations were submitted to NRC, 10 CFR 50.55a required KPS to follow the 1989 Edition of ASME B&PVC Section XI. At that time, the current reactor vessel integrity assessments satisfied the requirements of 10 CFR 50.60 with one exception. In Reference 4, Wisconsin Public Service Corporation (WPSC – former licensee) requested an exemption pursuant to 10 CFR 50.12 to use ASME B&PVC Code Cases N-514 and N-588. NRC approved use of ASME B&PVC Code Case N-588 in Reference 5. NRC authorized use of ASME B&PVC Code Case N-514 at KPS in License Amendment 144, dated April 1, 1999 (Reference 6).

Currently, 10 CFR 50.55a requires that KPS follow the 1998 Edition (through 2000 Addendum) of ASME B&PVC, Section XI. The 1998 Edition (through 2000 Addendum) of ASME B&PVC Section XI has incorporated the provisions of Code Case N-514 and N-588 into Appendix G. Thus, an exemption to the rule is not needed for the proposed heatup and cooldown limit curves and LTOP limitations with regard to enabling temperature and K_{Ic} (plane strain fracture toughness) methodology. However, the proposed heatup and cooldown limit curves and LTOP limitations for KPS are based upon utilization of NRC-approved exemptions from specific requirements of 10 CFR 50, Appendices G and H, and 10 CFR 50.61. The validity of these exemptions have not been affected by any subsequent KPS licensing actions and therefore the exemptions are considered to be currently applicable to KPS. Specifically, in Reference 2, subject to stated conditions, NRC approved:

1. An exemption to establish the use of a new methodology to meet the requirements of 10 CFR Part 50, Appendix G;
2. An exemption to modify the basis for the KPS reactor pressure vessel surveillance program (required by 10 CFR Part 50, Appendix H) to incorporate the acquisition of fracture toughness data; and
3. An exemption to establish the use of a new methodology to meet the requirements of 10 CFR 50.61.

The proposed heatup and cooldown limit curves and LTOP limitations were generated using a combination of:

1. The most limiting Adjusted Reference Temperature (ART) values;
2. NRC approved methodologies cited in Reference 2; and,
3. The "axial-flaw" and "circ-flaw" methodologies, which make use of the K_{Ic} methodology, included in the 1998 Edition (through 2000 Addendum) of ASME B&PVC Section XI, Appendix G.

For the weld wire heat 1P3571, the ART values are derived using the NRC-approved methodology for the master curve results. For the forging, the ART values are based on methods stated in 10 CFR 50.61, Regulatory Guide 1.99, Revision 2, and the 1998 Edition (through 2000 Addendum) of ASME B&PVC Section XI, Appendix G. The highest ART values correspond to the end of the expected extended license period (EOLE), at 52.1 EFPY, for the upper shell forging 123W250VA1 ("axial-flaw" orientation) and weld wire heat 1P3571 ("circ-flaw" orientation).

The proposed heatup and cooldown limit curves were created using a combination of both the "axial-flaw" and "circ-flaw" limiting ART values. The "circ-flaw" methodology, originating from Code Case N-588, is less restrictive than the standard "axial-flaw" methodology from the 1998 Edition (through 2000 Addendum) of ASME B&PVC Section XI. However, due to low ART values for the forgings, the "circ-flaw" is limiting in various portions of the proposed heatup and cooldown limit curves.

The proposed LTOP enabling temperature is based on the one quarter thickness (¼ T) ART value for weld wire heat 1P3571 and the methodology cited in the 1998 Edition (through 2000 Addendum) of ASME B&PVC Section XI, Appendix G.

A revised Pressurized Thermal Shock (PTS) evaluation, which forms the basis for the inside surface ART values of the circumferential beltline region weld, is based on implementation of the NRC-approved exemptions in Reference 2.

4.0 TECHNICAL ANALYSIS

4.1 Overview

The existing KPS P/T Limits and LTOP enabling temperature are valid through 31.1 EFPY. Reference 1 informed NRC that the KPS Reactor Pressure Vessel (RPV) is projected to reach 31.1 EFPY during operating cycle 31 (currently projected to occur approximately April 2012).

The following five Westinghouse Electric Company WCAP reports listed in Table 1 (and enclosed with this request) provide technical justification for this proposed amendment.

Table 1
Westinghouse Electric Company WCAP Reports

Enclosure	Identification	Title
1	WCAP-15074 Revision 1	Evaluation of the 1P3571 Weld Metal from the Surveillance Programs for Kewaunee and Maine Yankee
2	WCAP-16609-NP Revision 0	Master Curve Assessment of Kewaunee Power Station Reactor Vessel Weld Metal
3	WCAP-16641-NP Revision 0	Analysis of Capsule T from Dominion Energy Kewaunee Power Station Reactor Vessel Radiation Surveillance Program
4	WCAP-16642-NP Revision 0	Evaluation of Pressurized Thermal Shock for Kewaunee Power Station
5	WCAP-16643-NP Revision 2	Kewaunee Power Station Heatup and Cooldown Limit Curves for Normal Operation

The WCAP reports listed in Table 1 provide the following:

1. A previously NRC-approved methodology, based on the master curve approach, for assessment of material properties of the KPS reactor vessel;
2. A summary of the KPS 1P3571 weld metal surveillance capsule test results performed to date which bound operation during the license renewal period;
3. Documentation of standard and/or supplemental surveillance capsule fracture toughness testing results for the beltline 1P3571 weld metal and forging material both in the unirradiated and irradiated condition;
4. A summary of material properties for the extended beltline region;
5. A pressurized thermal shock (PTS) evaluation in accordance with the master curve method; and,

6. Heatup and cooldown limit curves corresponding to EOLE fluence along with the corresponding LTOP enabling temperature.

Reference 1 previously provided NRC with both WCAP-16641-NP, "Analysis of Capsule T from Dominion Energy Kewaunee Power Station Reactor Vessel Radiation Surveillance Program," and WCAP-16609-NP, "Master Curve Assessment of Kewaunee Power Station Reactor Vessel Weld Metal," to NRC. These WCAP reports are also enclosed with this proposed amendment for completeness.

NRC previously approved application of the Master Curve methodology in Reference 2. In Reference 3, DEK agreed to use the Master Curve Method as modified by the NRC staff and incorporate surveillance data from an additional capsule into the evaluation of the KPS reactor pressure vessel (RPV). In addition, DEK agreed in Reference 3 to obtain the following information regarding the KPS's reactor vessel radiation surveillance capsule:

1. A valid measurement of the fracture toughness-based T_0 parameter for the KPS RPV surveillance weld,
2. An estimate of the Charpy V-notch 30 ft-lb transition temperature shift for the surveillance weld, and
3. An estimate of the upper shelf energy (USE) drop for the surveillance weld.

Reference 1 transmitted the requested surveillance capsule data to the NRC. Enclosures 4 and 5 to this letter incorporate the surveillance capsule data into evaluations for the KPS RPV based upon the master curve method, as modified by the NRC staff.

The proposed P/T Limits and LTOP enabling temperature are derived using the same methods that were employed for the existing Technical Specifications except that:

1. The master curve method, as modified by the NRC staff, is used for assessing the material properties of the KPS reactor vessel beltline weld; and,
2. Fluence corresponds to the end of a (20 year) license extension (EOLE).

With these exceptions, the input parameters for evaluations of the KPS RPV are the same as those used for the existing Technical Specifications, although they have been revised to reflect higher fluence values corresponding to EOLE fluence. The existing TS were derived using ASME Code Case N-588 and ASME Code Case N-514. 10 CFR 50.55a requires KPS to follow the 1998 Edition (through 2000 Addendum) of ASME B&PVC Section XI. It is noted that the use of ASME Code Case N-588, previously approved as an exemption to the rule, has been incorporated into the 1998 Edition (through 2000 Addendum) of ASME B&PVC Section XI, Appendix G. Furthermore, the use of ASME Code Case N-514 has also been incorporated into the 1998 Edition of ASME B&PVC Section XI, Appendix G. In addition, the 1998 Edition (through 2000 Addendum) of ASME B&PVC Section XI, Appendix G uses K_{Ic} while the existing P/T Limits and LTOP enabling temperature were developed using K_{Ia} (the critical value of the stress intensity factor for crack arrest).

4.2 Determination of Chemistry Factors

10 CFR 50.61, "Fracture toughness requirements for protection against pressurized thermal shock events," and RG 1.99, Revision 2 identify the general procedures acceptable to the NRC for calculating the effects of neutron irradiation embrittlement of the low-alloy steels used in light-water-cooled reactor vessels. 10 CFR 50.61 and RG 1.99, Revision 2 describe two methods acceptable to the NRC for evaluating the predictions of radiation embrittlement needed to implement 10 CFR 50, Appendices G and H. 10 CFR 50.61, paragraph (c)(2)(ii)(A) requires that licensees determine a material-specific value of the chemistry factor (CF) when the surveillance data is deemed credible according to the criteria of 10 CFR 50.61, paragraph (c)(2)(i). If the surveillance data is not deemed credible, then the chemistry factor is determined using the copper and nickel content according to the criteria of 10 CFR 50.61, paragraph (c)(1)(iv)(A). Additionally, 10 CFR 50.61, paragraph (c)(2)(ii)(B) specifies if the chemical content of the surveillance weld differs from the average for the weld wire heat number associated with the vessel weld, then the measured values of ΔRT_{NDT} (the mean value of the adjustment in reference temperature caused by irradiation) must be adjusted for differences in copper and nickel content by multiplying them by the ratio of the chemistry factor for the vessel material and that of the surveillance weld.

The upper shell forging and circumferential weld are located in the beltline region of the KPS reactor vessel. The proposed heatup and cooldown limit curves have been constructed by combining the most conservative pressure temperature limits of the following limiting materials: upper shell forging, beltline weld, and vessel flange. This approach results in composite curves based on the most limiting material properties, projected flaw orientation, and fluence projections for the upper shell forging, vessel flange, and circumferential weld. The upper shell forging and vessel flange are the limiting materials in the low temperature region of the proposed heatup and cooldown limit curves. The circumferential weld is the limiting material in the higher temperature region of the proposed heatup and cooldown limit curves.

Both methods for establishing the chemistry factor have been utilized for the proposed heatup and cooldown limit curves, since the upper shell forging, vessel flange, and circumferential weld are limiting materials for establishing a single set of composite limit curves.

The surveillance capsule data for the circumferential weld meets the credibility criteria of 10 CFR 50.61, paragraph (c)(2)(i) and has been used for establishing the chemistry factor of the circumferential weld. There is no surveillance capsule data for the upper shell forging or for the vessel flange. Thus, the chemistry factor is taken from 10 CFR 50.61, Table 1.

Consistent with 10 CFR 50.61, paragraphs (c)(2)(ii)(A), (c)(1)(iv)(A), and (c)(2)(ii)(B), for purposes of evaluating the PTS of the KPS reactor vessel, DEK used plant specific surveillance capsule data to determine the actual shift in material properties. DEK has also applied a heat uncertainty term utilizing the ratio method to account for observed differences in shift between the KPS and Maine Yankee surveillance capsule test results. For purposes of developing the heatup and cooldown limit curves, DEK has used the copper and nickel content of the upper shell forging for determining the chemistry factor of the forging and the plant specific weld metal surveillance data to determine a material-specific value of the chemistry factor for the weld. However, the weld material used to fabricate the limiting circumferential weld in the KPS reactor vessel was also used for fabrication of one of the welds in the Maine Yankee reactor vessel.

4.3 Determination of RT_{NDT}

The heat-adjusted RT_{NDT} values of the circumferential weld have been derived from fracture toughness testing using the NRC-approved methodology for the master curve method. The most recent details regarding the fracture toughness testing are documented in WCAP-16641-NP, Revision 0 (Enclosure 3). Details regarding the NRC-approved methodology for the master curve method are documented in the Safety Evaluation Report (SER) contained in Reference 2. Details for application of the master curve method are documented in WCAP-16609-NP, Revision 0 (Enclosure 2).

To date, fracture toughness testing has been completed using specimens of circumferential weld metal obtained from three (3) surveillance capsules: Maine Yankee (MY) Capsule A-35, KPS Capsule S, and KPS Capsule T. WCAP-16642-NP (Enclosure 4), Table 11 documents that the heat-adjusted RT_{PTS} (RT_{NDT} evaluated for EOLE fluence) value for the circumferential weld metal corresponding to a fluence projection of 5.37 x 10^{19} n/cm^2 (52.1 EFPY) is 297.5 °F. A heat-adjusted RT_{PTS} value of 297.5 °F for the circumferential weld is lower than the PTS screening criteria of 300 °F, and is therefore acceptable. The ¼ T and ¾ T ART values for the circumferential weld metal used for the proposed heatup and cooldown Appendix G limit curves are derived from the heat-adjusted RT_{PTS} value of 297.5 °F and the attenuation factor specified in Regulatory Guide 1.99, Revision 2.

4.4 Determination of LTOP Enabling Temperature

The temperature restriction cited in KPS TS 3.1.a.1.C, TS 3.1.a.3, TS 3.1.b.1.C, and TS 3.1.b.4 is commonly referred to as the LTOP system enabling temperature. In accordance with Branch Technical Position BTP 5-2, "Overpressurization Protection of PWR's While Operating at Low Temperatures," this parameter has historically been defined as the water temperature corresponding to a metal temperature of at least RT_{NDT} + 90 °F at the beltline location (¼ T or ¾ T) that is controlling in the Appendix G limit calculations.

The revised temperature restriction is based on a new definition for enabling temperature, which is endorsed by NRC through 10 CFR 50.55a. The new enabling temperature is defined as (RT_{NDT} + 50 °F) in the 1998 Edition (through 2000 Addenda) of ASME B&PVC Section XI, Appendix G.

In addition, per Reference 2, ASME Code Case N-629 is used to establish a value of EOLE reference temperature (RT_{To}) in lieu of RT_{NDT}. Therefore, the new temperature restriction value, PTS assessment, and heatup and cooldown limit curves are calculated using the 1998 Edition (through 2000 Addenda) of ASME B&PVC Section XI, Appendix G, EOLE reference temperature (RT_{To}), and EOLE adjusted reference temperature ($ART_{To\text{-}EOLE}$) values. The values of $RT_{To\text{-}EOLE}$ and $ART_{To\text{-}EOLE}$ have been derived, in lieu of $RT_{PTS\text{-}EOLE}$ and ART_{EOLE}, using the methodology previously endorsed and approved by NRC in Reference 2.

4.5 Calculation of Appendix G Temperature and Pressure Limits

Maximum allowable pressures, reactor coolant temperature, and system heatup and cooldown rates have been determined for the KPS reactor vessel closure flange and beltline materials for hydrostatic pressure and leak tests, normal operation including anticipated operational occurrences, and low temperature overpressure protection (corresponding to isothermal events during low temperature operations (i.e., ≤ 343 °F)). These limitations have been calculated using the approved methodology described in 10 CFR 50.61 (as amended by Reference 2) and the 1998 Edition (through 2000 Addendum) of ASME B&PVC Section XI, Appendix G.

The calculation used the following inputs:

1. The neutron fluence (E > 1 MeV) values are based on projected operating hours through the EOLE period (December 21, 2033) utilizing a 95.6% capacity factor. EOLE corresponds to 52.1 EFPY.

   Due to power uprate and new fluence projections associated with surveillance capsule T, the existing heatup and cooldown limit curves and LTOP enabling temperature are currently being limited from 33 EFPY to 31.1 EFPY. The neutron fluence projection for the previous heatup and cooldown limit curves and LTOP enabling temperature at 31.1 EFPY is 3.34×10^{19} n/cm^2. The new neutron fluence projection at the vessel inner radius for the heatup and cooldown limit curves and LTOP enabling temperature is 5.37×10^{19} n/cm^2 (52.1 EFPY).

   The existing and proposed heatup and cooldown limit curves and LTOP enabling temperature have been derived using the latest ENDF/B-VI dosimetry cross sections. Furthermore, the neutron transport and dosimetry methodologies follow the guidance of Regulatory Guide 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence.” The KPS radiation surveillance program consists of six (6) capsules: V, R, P, T, S, and N. Four (4) capsules (V, R, P, and S) monitored the reactor vessel through EOL fluence. Capsule T monitored the reactor vessel to verify life extension material properties and fluence. Capsule N is an extra standby capsule. The existing heatup and cooldown limit curves and LTOP enabling temperature are based on radiation surveillance information from capsules V, R, P, and S. The revised enabling temperature and heatup and cooldown limit curves are based on radiation surveillance information from capsules V, R, P, S, and T.

2. 10 CFR 50.61,paragraph (c)(1)(iv)(A), which requires that licensees determine a chemistry factor using the best estimate weight percentage (Wt %) copper (Cu) and Wt % nickel (Ni).

   This paragraph is applicable to KPS for establishing the chemistry factor, since the surveillance data for the upper shell forging does not satisfy the credibility criteria. For the upper shell forging, which is controlling in the lower temperature region of the proposed heatup and cooldown limit curves, the weight percentage is 0.12 Wt % Cu and 0.71 Wt % Ni.

   WCAP-16643-NP (Enclosure 5), Table 2-5 provides a limiting chemistry factor value for the upper shell forging of 84.65 °F. Adjustment of the chemistry factor is not required, since the test specimens have been obtained from the upper shell forging used to fabricate the reactor vessel. Therefore, a chemistry factor value of 84.65 °F is used for the upper shell forging for the existing and proposed heatup and cooldown limits curves.

3. 10 CFR 50.61, paragraph (c)(2)(ii)(A), which specifies that the chemistry factor be determined from surveillance capsule data if the surveillance data satisfies the credibility criteria.

   The surveillance data for the beltline weld has been found to be credible. 10 CFR 50.61, paragraph (c)(2)(ii)(A) has been used for establishing the chemistry factor and chemistry factor ratio for the 1P3571 KPS weld metal. Use of the beltline weld properties, established in accordance with the NRC approved methodology in Reference 2, results in pressure and temperature limits that are more conservative at the upper end of the proposed heatup and cooldown curves. An adjusted chemistry factor value of 219.9 °F was used for the existing heatup and cooldown limit curves. WCAP-16642-NP (Enclosure 4), Table 6 documents the derivation of the beltline weld chemistry factor for the proposed heatup and cooldown curves. A beltline weld adjusted chemistry factor value of 219.17 °F has been derived in accordance with 10 CFR 50.61, paragraph (c)(2)(ii)(A) for the proposed heatup and cooldown curves. The difference in the beltline weld chemistry factor values for the existing and proposed heatup and cooldown curves is related to using new surveillance capsule data obtained from Capsule T.

4. The current heatup/cooldown limit curves, which are based on the methodology defined in Regulatory Guide 1.99, Revision 2 and NRC approval to use Code Case N-588.

The lower temperature portion of the existing cooldown P/T limit curves and heatup P/T limit curves (and corresponding LTOP system enabling temperature) is based on the following inputs and intermediate forging related data:

- Initial RT_{NDT} of 60 °F;
- 0.06 Wt % Cu and 0.71 Wt % Ni;
- Chemistry factor value of 37 °F; and,
- EOL fluence projected through 33 EFPY (but subsequently limited to 31.1 EFPY for power uprate considerations).

The lower temperature portion of the proposed heatup and cooldown P/T limit curves is based upon the following inputs and upper shell forging related data:

- Initial RT_{NDT} of 60 °F;
- 0.12 Wt % Cu and 0.71 Wt % Ni;
- Chemistry factor value of 84.65 °F; and,
- EOL fluence projected through 52.1 EFPY.

The higher temperature region of the existing heatup and cooldown P/T limit curves is based on the following inputs and circumferential weld related properties:

- Initial RT_{NDT} of -50 °F;
- An adjusted chemistry factor of 219.9 °F derived in accordance with 10 CFR 50.61, paragraph (c)(2)(ii)(A);
- A margin term of 28 °F; and,
- EOL fluence projected through 33 EFPY (but subsequently limited to 31.1 EFPY for power uprate considerations).

The higher temperature portion of the proposed heatup and cooldown P/T limit curves is based upon the following inputs derived from the previously NRC approved methodology related to the master curve approach for the circumferential weld:

- Chemistry factor of 219.17 °F;
- Total margin of 62.5 °F; ¼ T limiting "circ-flaw" ART of 278 °F;
- ¾ T limiting "circ-flaw" ART of 230 °F; and,
- EOLE fluence projected corresponding to 52.1 EFPY.

The horizontal and vertical portions of the proposed heatup and cooldown P/T limit curves, located between 73 °F to 193 °F, are limited by the reactor vessel flange unirradiated RT_{NDT} of 60 °F. 10 CFR 50, Appendix G specifies that the metal temperature of the closure flange regions must exceed the material unirradiated RT_{NDT} by at least 120 °F for normal operation and 90 °F for hydrostatic pressure tests and leak tests, when the pressure exceeds 20 percent of the preservice hydrostatic test pressure. In addition, when the core is critical, the pressure-temperature limits for core operation (except for low power physics tests) require that the reactor vessel be at a temperature equal to or higher than the minimum temperature required for the inservice hydrostatic test, and at least 40 °F higher than the minimum permissible temperature in the corresponding pressure-temperature curve for heatup and cooldown. These limits are incorporated into the pressure-temperature curves as applicable.

5. The current LTOP system enabling temperature, which is based on the definition provided in Code Case N-514.

The current LTOP enabling temperature is 200 °F. The intent of the current LTOP enabling temperature is to protect against exceeding the P/T curves during either a mass or energy input transient.

The current isothermal cooldown P/T limit curve (from 73 °F to 273 °F) and corresponding LTOP system enabling temperature are based on the following inputs and intermediate forging properties:

- Initial RT_{NDT} of 60 °F; 0.06 w/o Cu and 0.71 w/o Ni;
- Chemistry factor value of 37 °F derived in accordance with 10 CFR 50.61, paragraph (c)(1)(iv)(A);
- A margin term of 34 °F; and,
- EOL fluence projected through 33 EFPY (but subsequently limited to 31.1 EFPY for power uprate considerations).

The most limiting current P/T limit curve applicable to LTOP events is the isothermal cooldown limit curve, which is a composite curve representing the intermediate forging and closure head flange properties up to 273 °F, and circumferential beltline weld properties above this temperature. Application of Code Case N-588 to the circumferential beltline weld resulted in the intermediate forging and closure head flange being used for the derivation of the lower temperature portion of the current P/T limit cooldown curves. Although the closure head was replaced in 2004, the reactor vessel flange is forged from the same material as the original closure head flange.

The proposed LTOP enabling temperature is 343 °F. The new LTOP enabling temperature is based upon the criteria specified in Appendix G of ASME Section XI, 1998 Edition through 2000 Addenda. This criterion states that:

a) LTOP systems shall be effective at coolant temperatures less than 200 °F or at coolant temperatures corresponding to a reactor vessel metal temperature less than RT_{NDT} + 50 °F, whichever is greater.

b) LTOP systems shall limit the maximum pressure on the vessel to 100% of the pressure determined to satisfy Equation (1).

The new LTOP enabling temperature is derived by using a value of RT_{NDT} for the ¼ T position of the circumferential weld corresponding to 52.1 EFPY, which is 278 °F, plus 50 °F, plus 15 °F to compensate for the maximum potential difference between the coolant temperature and metal temperature.

The proposed Appendix G isothermal cooldown curve and the 5 °F/hour heatup curve are composite curves representing the upper shell forging, reactor vessel flange, and circumferential beltline weld properties. The proposed Appendix G heatup and cooldown curves include uncertainty for instrumentation.

Actuation of the LTOP system does not rely on instrumentation since it consists of a spring loaded relief valve. Therefore, in the future, a potential LTOP excursion will be compared against the proposed allowable pressure-temperature values for either the isothermal cooldown limit curve or the 5 °F /hr heatup limit curve, whichever is applicable, after correcting for instrumentation margin. The current setpoint for the LTOP relief valve, RHR-33-1, will remain unchanged at 500 psig.

6. In support of developing the proposed Appendix G P/T Limits and LTOP enabling temperature, RT_{NDT} and upper shelf energy (USE) values were also verified for each of the reactor vessel beltline and extended beltline materials with fluence projected to exceed 1 x 10^{17} n/cm^2 (E > 1.0 MeV).

    a) 10 CFR 50.61, paragraph (b)(2) indicates that the pressurized thermal shock (PTS) criterion is 270 °F for plates, forgings, and axial weld materials; and 300 °F for circumferential weld materials. RT_{NDT} values have been calculated for each of the reactor vessel beltline materials with projected fluence in excess of 1 x 10^{17} n/cm^2 (E > 1.0 MeV). WCAP-16642-NP (Enclosure 4), Tables 10 and 11 contain the RT_{PTS} calculations for the beltline and extended beltline region materials at EOLE (52.1 EFPY). The RT_{PTS} values for each of the beltline and extended beltline materials meet the PTS criterion through EOLE (52.1 EFPY).

    b) 10 CFR 50, Appendix G, paragraph IV.A.1.a states that reactor vessel beltline materials must maintain Charpy upper shelf energy (USE) throughout the life of the vessel of no less than 50 ft-lb. USE values have been measured or calculated for each of the reactor vessel materials with projected fluence in excess of 1 x 10^{17} n/cm^2 (E > 1.0 MeV). WCAP-16642-NP (Enclosure 4), Table A-2 provides a tabulation of the USE values for each of the beltline and extended beltline materials. The USE values for the beltline and extended beltline materials have been verified to remain above 50 ft-lb at EOLE (52.1 EFPY).

7. 10 CFR Part 50, Appendix H states that the purpose of the reactor vessel material surveillance program is to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region which result from exposure of

these materials to neutron irradiation and the thermal environment. This goal is accomplished by implementing ASTM E 185, "Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels." The practice of implementing ASTM E 185 is a two step process:

a) The design of the surveillance program and the capsule withdrawal schedule must meet the requirements of the edition of ASTM E 185 that is current on the date of the ASME Code to which the reactor vessel was purchased.
b) For each capsule withdrawal, the test procedures and reporting requirements must meet the requirements of ASTM E 185-82 to the extent practicable for the configuration of the specimens in the capsule.

The design of the KPS reactor vessel materials surveillance program is documented in KPS USAR section 4.4.1.1. The program consists of six (6) capsules. To date, five (5) capsules have been removed and tested to monitor changes in fracture toughness in reactor vessel beltline materials at fluence corresponding up to EOLE (52.1 EFPY). The sixth (6^{th}) capsule remains in the KPS reactor vessel as a standby capsule and may be used to project changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region should a decision be made to operate the plant beyond 60 years.

4.6 Heatup and Cooldown Limit Curves and LTOP System Enabling Temperature

The revised heatup and cooldown limit curves and LTOP enabling temperature have been developed to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests. The method used in preparing the heatup and cooldown limit curves is presented in WCAP-16643-NP (Enclosure 5). Supporting information can be found in Reference 2 and Enclosures 1, 2, 3, 4, and 5. The method used for evaluation of the beltline weld metal, upper shell forging, and reactor vessel flange is consistent with ASME B&PVC Section XI, Appendix G, NRC Standard Review Plan Chapter 5.3.2, 10 CFR 50.61, and the NRC SER documented in Reference 2.

The safety factors and margins applied in the preparation of the heatup/cooldown P/T limit curves and LTOP enabling temperature meet the criteria set forth by these documents.

Inherent conservatisms were used in the development of the P/T limits and include:

1. An assumed defect in the reactor vessel wall with a depth equal to 1/4 of the thickness of the vessel wall (1/4T) and a length equal to 1.5 times the thickness of the vessel wall.
2. An assumed reference flaw oriented in both axial and circumferential directions applied to the upper shell forging. With incorporation of Code Case N-588 into the

1998 Edition (through 2000 Addenda) of ASME B&PVC Section XI, Appendix G an assumed reference flaw oriented in the circumferential direction is imposed on the circumferentially oriented beltline weld. At KPS, the only weld in the core region is oriented in the circumferential direction.

3. A safety factor of 2 is applied to the membrane stress intensity factor.
4. The limiting toughness is based upon a reference value (KIC), which is a lower bound on the static fracture toughness.
5. A margin of 30 psi and 13 °F is used to account for plant instrumentation uncertainty.
6. A pressure margin of 70 psi is used for the difference between the instrument and beltline region.
7. A 2-sigma margin term of 34 °F is applied in determining the adjusted reference temperature for the upper shell forging.
8. An implicit margin on the order of 2 °F in the NRC SER methodology is used to derive RT_{PTS} for the circumferential weld metal.
9. An implicit margin of 28 °F for $\sigma_{\Delta T_o}$ to account for uncertainty in the irradiation embrittlement behavior of the KPS RPV circumferential weld.
10. A bounding value of 14 °F for σ_{T_o} to account for uncertainty in initial properties for the KPS RPV circumferential weld.
11. A bias value of 8.5 °F for each of the Precracked Charpy V-notch (PCVN) based T_o values relative to the values obtained with larger size compact tension (CT) specimens for the KPS RPV circumferential weld.

Since the LTOP enabling temperature is based upon the upper portion of the proposed P/T isothermal cooldown limit and 5 °F/hr heatup limit curves, which are based on the beltline weld, the above margins equally apply to the LTOP enabling temperature. Beyond the conservatism described above, DEK has incorporated the following additional margin into the proposed heatup and cooldown limit curves and LTOP enabling temperature:

1. An instrumentation margin of 16.6 psig is included in the LTOP setpoint determination.
2. Use of very conservative heat transfer coefficients (7000 Btu/ hr-ft^2-°F), and neglecting the effects of cladding conductivity in the analysis of thermal stress in the P/T curves. Furthermore, the thermal stress is calculated for fixed and constant rates of temperature change and does not reflect the intermittent rates actually experienced by the vessel. That is, hold points for evolutions such as crud burst cleanup during shutdown and securing residual heat removal from service during startup act as a thermal soak period and reduce the integrated effect on thermal stress.

The following additional conservatisms were not factored into the development of the proposed amendment:

1. Although no flaws exceeding the ASME Section XI allowable flaw size for volumetric examination have been detected during any of the three inservice examinations of the reactor vessel (which were performed in 1985, 1995, and 2004), the assumption of a 1/4 thickness surface flaw was nonetheless maintained.
2. A 2-inch diameter spring loaded safety valve, set at 480 psig, is located in the LTOP/RHR system. At 500 psig, the LTOP relief valve setpoint, the relieving capacity of this smaller valve is 230 gpm.
3. The actual LTOP relief valve capacity is at least 10% greater than the capacity used in the design and setpoint analyses. This is in accordance with the requirements of ASME Section III, NC-7000.
4. Assumptions in the overpressure transient analyses are conservative relative to the actual Kewaunee reactor coolant system (RCS) and design and operating practices:
   a) The RCS was assumed to be rigid with respect to metal expansion.
   b) No credit was taken for the shrinkage effect caused by low temperature safety injection water added to higher temperature reactor coolant.
   c) No credit was taken for the reduction in reactor coolant bulk modulus at RCS temperatures above 100 °F (constant bulk modulus at all RCS temperatures).
   d) The entire volume of water of the steam generator secondary side was assumed available for heat transfer to the primary side. In reality, the liquid immediately adjacent and above the tube bundle would be the primary source of energy in the transient.
   e) The overall steam generator heat transfer coefficient, U, was assumed to be the free convective heat transfer coefficient of the secondary side, h_{sec}. The forced convective heat transfer coefficient of the primary side, h_{pn}, and the tube metal resistance has been ignored, thus resulting in a conservative (high) coefficient.
   f) The reactor coolant pump start time assumed in the heat input analysis was 9-10 seconds; whereas, the Kewaunee pump startup time is approximately 20 seconds.

Due to the capabilities of NDE, it is considered very unlikely that large flaws would remain undetected in the beltline region of reactor vessels. It is also unlikely that axial cracks originating from a circumferential weld perpendicular to the weld seam orientation would remain undetected in reactor vessels. Both industry experience and engineering studies indicate that the primary degradation mechanism affecting the beltline region of the reactor vessel is neutron embrittlement. No other service-induced degradation mechanism is known to exist in a pressurized water reactor which might cause a prior existing defect located in the beltline region of the reactor vessel to grow while in service. Based on these considerations, and the fact that the pressure temperature limit for reactor operation is the limiting pressure for any of the materials in the vessel, it is not necessary to include additional conservatism in the assumed flaw orientation for circumferential welds. ASME Section XI Code Case N-588 and the accompanying Appendix G Code change corrects this inconsistency in assumed flaw orientation for circumferential welds in vessels when calculating operating P-T limits.

During development of Code Case N-588 and the accompanying Code change, the ASME Section XI Working Group Operating Plant Criteria (WGOPC) developed a solution of stress intensity factors for inside and outside surface reference flaws, performed a circumferential flaw margin assessment, and provided justification of the Code change. Thus, the assumed flaw orientation for the beltline weld is based upon the current requirements specified in 1998 Edition through 2000 Addendum of the ASME Code, Section XI, Appendix G.

10 CFR 50.60(a) states that the reactor coolant pressure boundary must meet the fracture toughness and material surveillance program requirements set forth in 10 CFR 50, Appendices G and H. 10 CFR 50.60(b) states that proposed alternatives to the described requirements in 10 CFR 50, Appendices G and H, or portions thereof, may be used when an exemption is granted by the Commission under 10 CFR 50.12. As stated herein, this proposed amendment would implement a methodology approved by NRC in Reference 2 for the master curve method.

Consistent with 10 CFR 50.60, DEK has measured the actual unirradiated and irradiated fracture toughness of heat 1P3571, the surveillance weld metal, at EOL and EOLE fluence. Fracture toughness testing was performed on weld specimens obtained from Capsules S and T of the KPS radiation surveillance program and Capsule A-35 of the Maine Yankee radiation surveillance program. The resulting fracture toughness data, which satisfies the requirements of ASTM E I921-97, has been used to determine reference temperature and adjusted reference temperature values for the beltline weld with margins similar to those of 10 CFR 50.61. The weld metal fracture toughness results along with application of the master curve method have been investigated and demonstrate that the methods used for this assessment of the KPS reactor vessel are conservative.

Five surveillance capsules have been removed and tested from the KPS reactor vessel. The surveillance capsule data has been evaluated to the five credibility requirements, originally specified in Regulatory Guide 1.99, Revision 2, and now incorporated into 10 CFR 50.61. This evaluation finds that the surveillance data for the circumferential beltline weld metal are credible. There is no surveillance data for the upper shell forging. Thus, Regulatory Position C.1 of Regulatory Guide 1.99, Revision 2 was used for calculating the chemistry factor of the reactor vessel upper shell forging and vessel flange material. Regulatory Position C.2 was used to calculate the chemistry factor and chemistry factor ratio for the beltline weld.

Use of a low leakage core design during Fuel Cycle 16 and all subsequent fuel cycles decreases the rate of shift in transition temperature from ductile to non-ductile behavior.

Compliance with 10 CFR 50.61 by use of 1998 Edition (through 2000 Addendum) of the ASME Code, Section XI, Appendix G and an NRC approved methodology for the master curve method is an acceptable approach for evaluating predictions of radiation embrittlement needed to implement 10 CFR 50, Appendices G and H. The resulting P/T limits meet the NRC acceptance criteria for the LTOP setpoint and system design as described in an NRC SER to WPSC dated September 6, 1985, which concluded that "the spectrum of postulated pressure transients would be mitigated... such that the temperature pressure limits of Appendix G to 10 CFR 50 are maintained." Radiological off-site exposure from postulated low temperature over pressurization events will not exceed regulatory requirements. Therefore, utilization of these curves will not adversely impact the consequences of any of the accidents in the Kewaunee USAR.

Based on the information above, the preparation of the revised heatup and cooldown limit curves and LTOP system enabling temperature meets the applicable safety criteria and regulatory guidance and therefore does not represent a safety concern.

4.7 Conclusion

The proposed P/T and LTOP limitations have been developed and determined in accordance with NRC approved methodology cited in 10 CFR 50, Appendices G and H, and 10 CFR 50.61; NRC approved exemptions to the rule in Reference 2; and, Appendix G of ASME Section XI, 1998 Edition (through 2000 Addendum).

The LTOP enabling temperature is being increased from 200 °F to 343 °F. By system design, the LTOP system is enabled whenever the residual heat removal (RHR) system is aligned with the suction side isolation valves to the reactor coolant system (RCS) open. TS 3.1.b.4 is being revised to require that LTOP be operable whenever one or more of the RCS cold leg temperatures are ≤ 343 °F and the reactor vessel head is installed. This change will thus require that RHR be aligned to provide the LTOP function when RCS temperature decreases to the LTOP enabling temperature. Factoring in instrument uncertainty (13 °F), LTOP will need to be enabled when indicated RCS temperature decreases to 356 °F.

The setpoint for the LTOP relief valve is unaffected by this proposed amendment and remains at 500 psig, which is adequate to mitigate the effects from either the existing postulated energy or mass addition events. Each of the reactor vessel beltline and extended beltline materials has RT_{PTS} values and upper shelf energies that satisfy regulatory criterion for neutron exposure projections through 52.1 EFPY. For the period of operation through 52.1 EFPY, changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region have been monitored by surveillance capsule T and used to establish the proposed P/T and LTOP limitations. The remaining surveillance capsule is adequate for monitoring fluence during the 52.1 EFPY period of operation.

The proposed heatup and cooldown P/T limit curves, along with the corresponding Technical Specification changes, are provided in Attachment 2. Attachment 3 contains marked up Technical Specification Bases pages. WCAP-15074, Revision 1 (Enclosure 1), WCAP-16609-NP, Revision 0 (Enclosure 2), WCAP-16641-NP, Revision 0 (Enclosure 3), WCAP-16642-NP, Revision 0 (Enclosure 4), and WCAP-16643-NP, Revision 2 (Enclosure 5) provide supporting documentation for the proposed heatup and cooldown limit curves, a revised PTS evaluation, and LTOP enabling temperature.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

The proposed amendment would revise the Kewaunee Power Station (KPS) heatup and cooldown limit curves and low temperature overpressure protection (LTOP) requirements. The propose amendment would change the KPS Technical Specifications (TS) associated with these parameters. The proposed TS changes include replacing the existing heatup and cooldown curves with new updated curves and incorporating a higher LTOP enabling temperature. The proposed amendment would extend the applicability period of the heatup and cooldown limit curves and LTOP limitations from the currently existing 31.1 effective full power years (EFPY) to the proposed value of 52.1 EFPY. Dominion Energy Kewaunee (DEK) has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?**

   No.

   The proposed amendment does not impact the capability of the reactor coolant pressure boundary (i.e., there is no change in operating pressure, materials, seismic loading, etc.). Specifically, design provisions and continued compliance with 10 CFR 50, Appendices G and H, are intended to ensure that failure of a reactor vessel is not a credible event. Therefore, the proposed change provides adequate design assurance, consistent with the provisions of 10 CFR 50, that reactor vessel and coolant system integrity will be maintained and that there is no increase in the potential for the occurrence of a loss of coolant accident (LOCA).

   The revised LTOP enabling temperature and revised pressure/temperature (P/T) limits continue to ensure that the 10 CFR 50, Appendix G P/T limits are not exceeded; and therefore, help ensure that reactor coolant system (RCS) integrity is maintained. The proposed amendment does not modify the reactor coolant system pressure boundary, or make any physical changes to the facility design, material, construction standards, or setpoints. The RCS operating pressure is not being changed. The LTOP setpoint also remains unchanged. The revised LTOP enabling temperature results in the LTOP function being effective over a wider temperature range, which is conservative. The probability of an LTOP event occurring is independent of the P/T limits for the RCS pressure boundary and enabling temperature. Therefore, the probability of a LTOP event is not increased.

   The revised heatup and cooldown limit curves and LTOP enabling temperature were developed using test results from specimens that represent the most limiting materials in the reactor coolant pressure boundary, which provides assurance that reactor pressure vessel fracture toughness requirements are met and the integrity of the RCS pressure boundary is maintained. The extended applicability period of the heatup and cooldown limit curves and LTOP limitations has no impact on the consequences of a postulated accident. The metallurgical properties of the reactor coolant pressure boundary will continue to comply with applicable regulatory and ASME Code requirements during the extended applicability period. The revised limit curves continue to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, thus ensuring the integrity of the reactor coolant pressure boundary through the extended applicability period. Therefore, this extending the period of the limit curves and LTOP limitations does not significantly impact the probability of previously evaluated accidents.

The proposed amendment does not adversely affect the integrity of the RCS, or any fission product barrier, such that their functions in the control of radiological consequences are affected. The changes do not degrade or prevent the response of the LTOP relief valve or other safety-related systems to mitigate previously evaluated accidents. In addition, the changes do not alter any assumption previously made in the radiological consequence evaluations nor affect the mitigation of the radiological consequences of an accident previously evaluated.

Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?**

   No.

   The revised LTOP system enabling temperature and Appendix G P/T limitations were prepared using methods derived from the ASME Boiler and Pressure Vessel Code and the criteria set forth in NRC Standard Review Plan 5.3.2. The proposed heatup and cooldown limit curves were conservatively developed by using the limiting material properties of the reactor coolant pressure boundary to form a single set of composite curves. These curves are based on the more limiting material properties associated with neutron exposure projections through 52.1 EFPY (rather than the previous 33 EFPY). These changes do not cause the initiation of any accident or create any new credible limiting failure for safety-related systems and components. The proposed amendment will not result in any event previously deemed incredible being made credible. As such, the proposed amendment does not create the possibility of an accident different than the accidents that have been previously evaluated.

   The proposed amendment does not involve any physical change to structures, systems, or components (SSCs). Therefore, the proposed amendment does not have any adverse effect on the ability of safety-related SSCs to perform their intended safety functions. The use of an extended applicability period for the heatup and cooldown limit curves and LTOP limitations does not modify the reactor coolant pressure boundary, or make any physical changes to the LTOP setpoint or design. Therefore, the proposed amendment does not create any new failure modes.

The revised LTOP operating band is within the existing design limits of the residual heat removal system, which provides the LTOP function. As such, there is no adverse impact on LTOP.

Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does the proposed amendment involve a significant reduction in a margin of safety?**

No.

The proposed Appendix G P/T limitations and LTOP enabling temperature were prepared using methods derived from the ASME Boiler and Pressure Vessel Code and the criteria set forth in NRC Standard Review Plan 5.3.2. Adherence to these criteria, along with the calculational limitations specified in 10 CFR 50.61 for continued reactor coolant system integrity, ensure that proper limits and safety factors are maintained. Specifically, the regulatory requirements for fracture toughness of ferritic materials in the pressure-retaining components of the reactor coolant pressure boundary will continue to be met with the proposed changes.

The revised heatup and cooldown limit curves and LTOP system enabling temperature were prepared using measurements of fracture toughness parameters for the limiting materials in the reactor coolant pressure boundary based on neutron exposure projections through 52.1 EFPY. The safety factors and margins used in the development of the limit curves and LTOP system enabling temperature meet the criteria set forth in the ASME Code and NRC regulations through the extended applicability period for the heatup and cooldown limit curves.

The revised limit curves and LTOP enabling temperature maintain adequate margins of safety during any condition of normal operation (including anticipated operational occurrences and system hydrostatic tests) and low temperature overpressure protection. The required fracture toughness of the reactor pressure vessel is maintained throughout the extended applicability period of the heatup and cooldown limit curves.

The proposed change will have no adverse affect on the availability, operability, or performance of safety-related systems and components. The ability of plant structures, systems, and components to perform their designated safety function is unaffected by this proposed amendment.

Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

Based on the above, DEK concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

10 CFR 50.60 provides acceptance criteria for fracture prevention measures for light water nuclear power reactors for normal operation.

10 CFR 50.60(a) states that the reactor coolant pressure boundary must meet the fracture toughness and material surveillance program requirements set forth in 10 CFR 50, Appendices G and H.

10 CFR 50.60(b) states that proposed alternatives to the described requirements in 10 CRR 50, Appendices G and H, or portions thereof, may be used when an exemption is granted by the Commission under 10 CFR 50.12.

10 CFR 50.61 provides fracture toughness requirements for protection against pressurized thermal shock (PTS) events.

10 CFR 50.61(b)(1) states that "for each pressurized water nuclear power reactor..., the licensee shall have projected values of RT_{PTS}, accepted by the NRC, for each reactor vessel beltline material for the EOL fluence of the material. The assessment of RT_{PTS} must use the calculation procedures given in paragraph (c)(1) of this section, except as provided in paragraphs (c)(2) and (c)(3) of this section. The assessment must specify the bases for the projected value of RT_{PTS} for each vessel beltline material, including the assumptions regarding core loading patterns, and must specify the copper and nickel contents and the fluence value used in the calculation for each beltline material. This assessment must be updated whenever there is a significant change in projected values of RT_{PTS}, or upon request for a change in the expiration date for operation of the facility."

10 CFR 50.61(b)(2) states that the pressurized thermal shock (PTS) screening criterion is 270 °F for plates, forgings, and axial weld materials, and 300 °F for circumferential weld materials. For the purpose of comparison with this criterion, the value of RT_{PTS} for the reactor vessel must be evaluated according to the procedures of paragraph (c) of this section, for each weld and plate, or forging, in the reactor vessel beltline. RT_{PTS} must be determined for each vessel beltline material using the EOL fluence for that material.

10 CFR 50, Appendix G to, "Fracture Toughness Requirements," specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the

reactor coolant pressure boundary of light water nuclear power reactors to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime. 10 CFR 50, Appendix G, Section IV provides fracture toughness requirements.

10 CFR 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements," specifies that the purpose of the material surveillance program required by this appendix is to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of light water nuclear power reactors which result from exposure of these materials to neutron irradiation and the thermal environment. Under the program, fracture toughness test data are obtained from material specimens exposed in surveillance capsules, which are withdrawn periodically from the reactor vessel. These data will be used as described in 10 CFR 50, Appendix G, Section IV. 10 CFR 50, Appendix H, Section III provides surveillance program criteria.

The basis for the proposed amendment includes exemptions previously granted by the Commission under 10 CFR 50.12. These include exemptions from specific requirements of 10 CFR 50, Appendices G and H, and from specific requirements of 10 CFR 50.61. Specifically, the NRC approved the following exemptions for KPS:

1. An exemption to establish the use of a new methodology to meet the requirements of 10 CFR 50, Appendix G;
2. An exemption to modify the basis for the KPS reactor pressure vessel surveillance program (required by 10 CFR 50, Appendix H) to incorporate the acquisition of fracture toughness data; and,
3. An exemption to establish the use of a new methodology to meet the requirements of 10 CFR 50.61.

Based on the considerations discussed above:

1. There is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner;
2. Such activities will be conducted in compliance with the Commission's regulations; and,
3. The issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 REFERENCES

1. Letter from David A. Christian (Dominion Generation) to Document Control Desk, "Dominion Energy Kewaunee, Inc. Kewaunee Power Station Irrradiated Reactor Vessel Surveillance Capsule Test Results for Kewaunee Capsule T Per 10 CFR 50 Appendix H," dated November 14, 2006. [ADAMS Accession Nos. ML063250414 and ML063250421]
2. Letter from John G. Lamb (NRC) to Mark Reddemann (NMC), "Kewaunee Nuclear Power Plant – Exemption from the Requirements of 10 CFR Part 50, Appendix G, Appendix H, and Section 50.61 (TAC NO. MA8585)," dated May 1, 2001. [ADAMS Accession No. ML011210180]
3. Letter from Kyle A. Hoops (NMC) to Document Control Desk (NRC), "Request for Exemption from the Requirements of 10 CFR Part 50 Appendixes G and H, and 10 CFR 50.61 (Master Curve)," dated March 12, 2001. [ADAMS Accession No. ML010800019]
4. Letter from M.L. Marchi (WPSC) to Document Control Desk (NRC), "Request for 10 CFR 50.60 Exemption – Use of ASME Code Cases N-514 & N-588," dated August 6, 1998.
5. Letter from William O. Long (NRC) to M. L. Marchi (WPSC), "Issuance of Exemption from 10 CFR 50.60 by Applying ASME Code Case N-588 for Kewaunee Nuclear Power Plant (TAC No MA2471)," dated November 25, 1998. [ADAMS Accession No. ML020770294]
6. Letter from William O. Long (NRC) to M. L. Marchi (WPSC), "Amendment No. 144 to Facility Operating License No. DPR-43, Kewaunee Nuclear Power Plant (TAC No. MA2284)," dated April 1, 1999. [ADAMS Accession No. ML020770334]
7. NRC Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2, dated May 1988.

ATTACHMENT 2

LICENSE AMENDMENT 246
REACTOR COOLANT SYSTEM PRESSURE TEMPERATURE LIMITS AND LOW TEMPERATURE OVERPRESSURE PROTECTION

MARKED UP TECHNICAL SPECIFICATIONS PAGES:

CTS 3.1 Reactor Coolant System
(including List of Figures and Figure TS 3.1-1 and Figure TS 3.1-2)

KEWAUNEE POWER STATION
DOMINION ENERGY KEWAUNEE, INC.

LIST OF FIGURES

~~Note:~~

~~[1] The curves are limited to 31.1 EFPY due to changes in vessel fluence associated with operation at uprated power.~~

3.1 REACTOR COOLANT SYSTEM

APPLICABILITY

Applies to the OPERATING status of the Reactor Coolant System (RCS).

OBJECTIVE

To specify those LIMITING CONDITIONS FOR OPERATION of the Reactor Coolant System which must be met to ensure safe reactor operation.

SPECIFICATIONS

a. Operational Components

1. Reactor Coolant Pumps

   A. At least one reactor coolant pump or one residual heat removal pump shall be in operation when a reduction is made in the boron concentration of the reactor coolant.

   B. When the reactor is in the OPERATING mode, except for low power tests, both reactor coolant pumps shall be in operation.

   C. A reactor coolant pump shall not be started with one or more of the RCS cold leg temperatures ≤ ~~200~~343°F unless the secondary water temperature of each steam generator is < 100°F above each of the RCS cold leg temperatures.

2. Decay Heat Removal Capability

   A. At least two of the following four heat sinks shall be OPERABLE whenever the average reactor coolant temperature is ≤ 350°F but > 200°F.

      1. Steam Generator 1A
      2. Steam Generator 1B
      3. Residual Heat Removal Train A
      4. Residual Heat Removal Train B

      If less than the above number of required heat sinks are OPERABLE, then corrective action shall be taken immediately to restore the minimum number to the OPERABLE status.

3.1.a.3 Pressurizer Safety Valves

LCO 3.1.a.3 Two pressurizer safety valves shall be OPERABLE

APPLICABILITY: Reactor Coolant System Temperature Greater than the Low Temperature Overpressure Protection (LTOP) Enabling Temperature (~~200~~343°F)

ACTIONS

- NOTE -

During a hydro test of the RCS, the pressurizer safety valves may be blanked provided the power-operated relief valves and the safety valve on the discharge pump are set for the test pressure plus 35 psi to protect the system.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pressurizer safety valve inoperable	A.1 Restore to OPERABLE status	15 Minutes
	OR	
	A.2 Be in HOT SHUTDOWN	12 Hours
B. Both pressurizer safety valves are inoperable	B.1 Restore one pressurizer safety valve to an OPERABLE status	15 Minutes
	OR	
	B.2 Be in a condition with the LTOP system OPERABLE or reactor vessel head removed	48 Hours

b. Heatup and Cooldown Limit Curves for Normal Operation

1. The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures TS 3.1-1 and TS 3.1-2. Figures TS 3.1-1 and TS 3.1-2 are applicable for the service period of up to ~~33[1]~~52.1 effective full-power years.

   A. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation.

   B. Figures TS 3.1-1 and TS 3.1-2 define limits to assure prevention of non-ductile failure only. For normal operation other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

   C. The isothermal curve in Figure TS 3.1-2 and 5°F/hr curve in Figure TS 3.1-1 define~~s~~ limits to assure prevention of non-ductile failure applicable to low temperature overpressurization events~~-only~~. Application of ~~this~~ these curves is limited to evaluation of LTOP events whenever one or more of the RCS cold leg temperatures are less than or equal to the LTOP enabling temperature of ~~200~~343°F.

2. The secondary side of the steam generator must not be pressurized > 200 psig if the temperature of the steam generator is < 70°F.

3. The pressurizer cooldown and heatup rates shall not exceed 200°F/hr and 100°F/hr, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is > 320°F.

4. The overpressure protection system for low temperature operation shall be OPERABLE whenever one or more of the RCS cold leg temperatures are ≤ ~~200~~343°F, and the reactor vessel head is installed. The system shall be considered OPERABLE when at least one of the following conditions is satisfied:

   A. The overpressure relief valve on the Residual Heat Removal System (RHR 33-1) shall have a set pressure of ≤ 500 psig and shall be aligned to the RCS by maintaining valves RHR 1A, 1B, 2A, and 2B open.

      1. With one flow path inoperable, the valves in the parallel flow path shall be verified open with the associated motor breakers for the valves locked in the off position. Restore the inoperable flow path within five days or complete depressurization and venting of the RCS through a ≥ 6.4 square inch vent within an additional eight hours.

      2. With both flow paths or RHR 33-1 inoperable, complete depressurization and venting of the RCS through at least a 6.4 square inch vent pathway within eight hours.

~~[1] The curves are limited to 31.1 EFPY due to changes in vessel fluence associated with operation at uprated power.~~

FIGURE TS 3.1-1
KEWAUNEE POWER STATION UNIT NO. 1 HEATUP LIMITATION CURVES
APPLICABLE FOR PERIODS UP TO 52.1 EFFECTIVE FULL-POWER YEARS

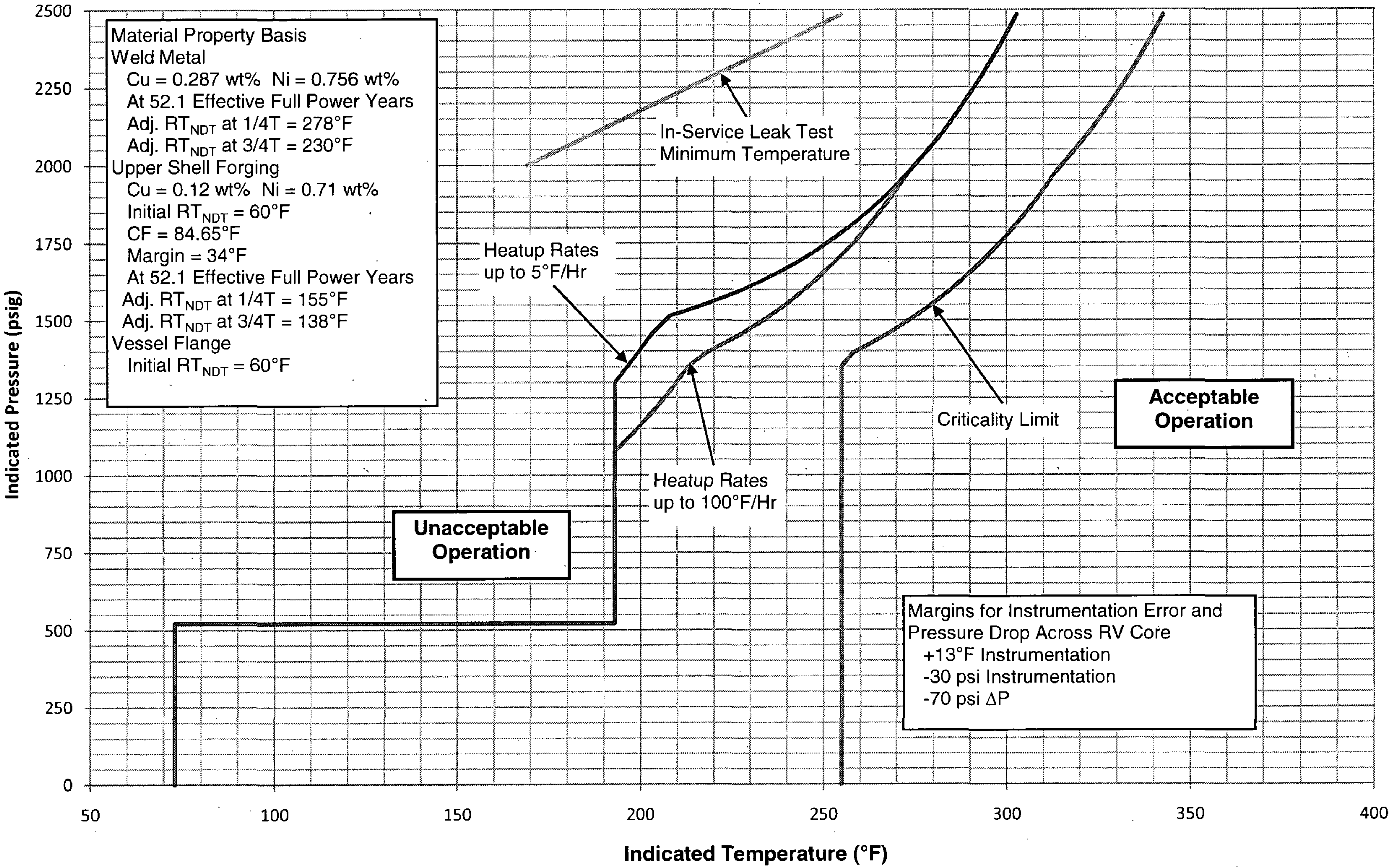
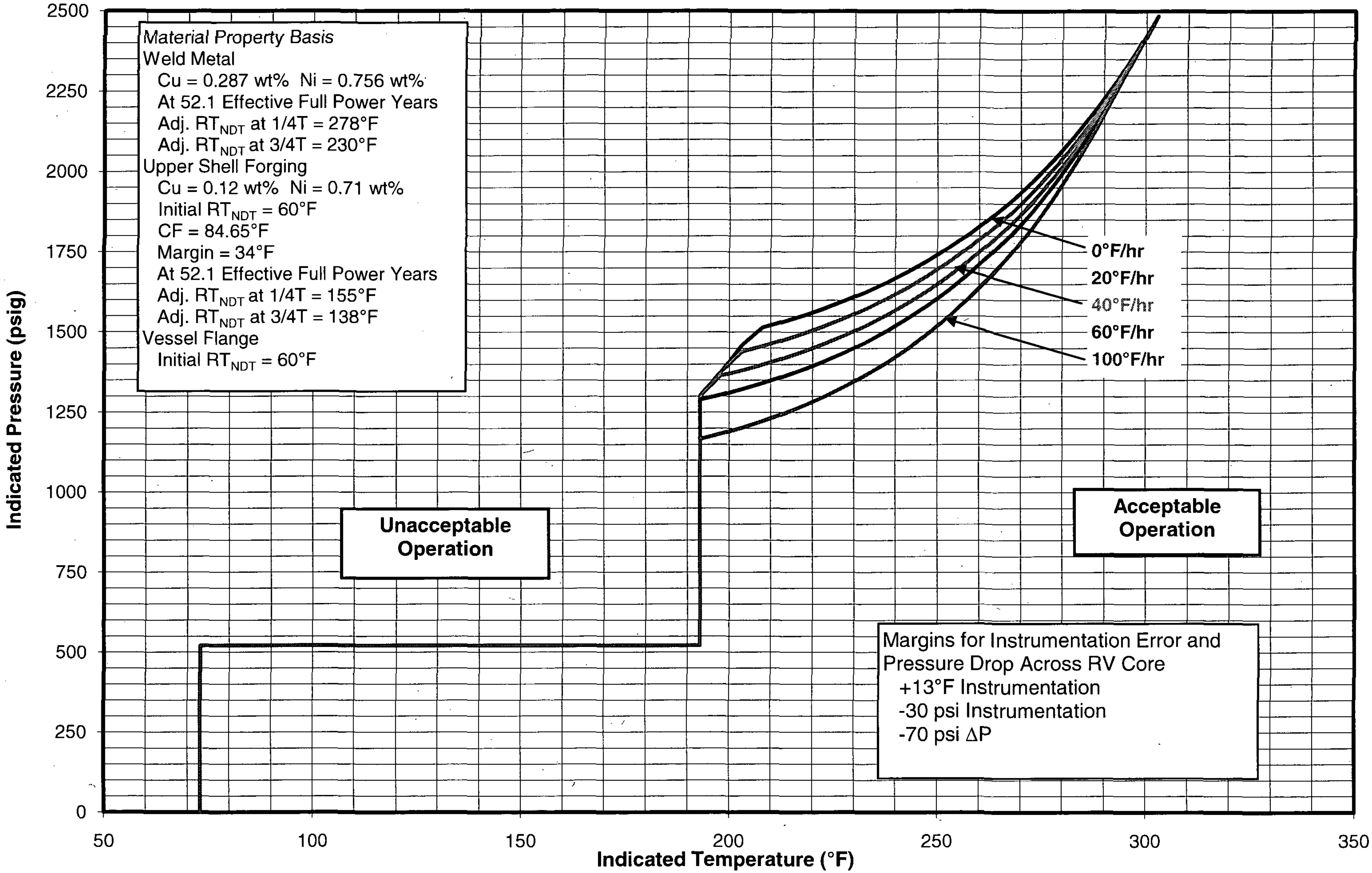


FIGURE TS 3.1-2
KEWAUNEE POWER STATION UNIT NO. 1 COOLDOWN LIMITATION CURVES
APPLICABLE FOR PERIODS UP TO 52.1 EFFECTIVE FULL-POWER YEARS



ATTACHMENT 3

LICENSE AMENDMENT 246
REACTOR COOLANT SYSTEM PRESSURE TEMPERATURE LIMITS AND LOW TEMPERATURE OVERPRESSURE PROTECTION

MARKED-UP TECHNICAL SPECIFICATIONS BASES PAGES:

CTS Basis – Reactor Coolant System (TS 3.1)

KEWAUNEE POWER STATION
DOMINION ENERGY KEWAUNEE, INC.

BASIS - Reactor Coolant System (TS 3.1.a)

Reactor Coolant Pumps (TS 3.1.a.1)

When the boron concentration of the Reactor Coolant System is to be reduced, the process must be uniform to prevent sudden reactivity changes in the reactor. Mixing of the reactor coolant will be sufficient to maintain a uniform boron concentration if at least one reactor coolant pump or one residual heat removal pump is running while the change is taking place. The residual heat removal pump will circulate the equivalent of the primary system volume in approximately one-half hour.

Part one of the specification requires that both reactor coolant pumps be OPERATING when the reactor is in power operation to provide core cooling. Planned power operation with one loop out-of-service is not allowed in the present design because the system does not meet the single failure (locked rotor) criteria requirement for this MODE of operation. The flow provided in each case in part one will keep Departure from Nucleate Boiling Ratio (DNBR) well above the DNBR limit. Therefore, cladding damage and release of fission products to the reactor coolant will not occur. One pump operation is not permitted except for tests. Upon loss of one pump below 10% full power, the core power shall be reduced to a level below the maximum power determined for zero power testing. Natural circulation can remove decay heat up to 10% power. Above 10% power, an automatic reactor trip will occur if flow from either pump is lost.[11]

The RCS will be protected against exceeding the design basis of the Low Temperature Overpressure Protection (LTOP) System by restricting the starting of a Reactor Coolant Pump (RXCP) to when the secondary water temperature of each SG is < 100°F above each RCS cold leg temperature. The restriction on starting a reactor coolant pump (RXCP) when one or more RCS cold leg temperatures is ≤ ~~200~~343°F (356°F indicated to account for instrument uncertainty) is provided to prevent a RCS pressure transient, caused by an energy addition from the secondary system, which could exceed the design basis of the LTOP System.

Decay Heat Removal Capabilities (TS 3.1.a.2)

When the average reactor coolant temperature is ≤ 350°F a combination of the available heat sinks is sufficient to remove the decay heat and provide the necessary redundancy to meet the single failure criterion.

When the average reactor coolant temperature is ≤ 200°F, the plant is in a COLD SHUTDOWN condition and there is a negligible amount of sensible heat energy stored in the Reactor Coolant System. Should one residual heat removal train become inoperable under these conditions, the remaining train is capable of removing all of the decay heat being generated.

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[1] USAR Section 7.2.2

Heatup and Cooldown Limit Curves for Normal Operation (TS 3.1.b)

Fracture Toughness Properties (TS 3.1.b.1)

The fracture toughness properties of the ferritic material in the reactor coolant pressure boundary are determined in accordance with the ASME Boiler and Pressure Vessel Code,[5] ~~and~~ the calculation methods of Footnote[6], and master curve methodology[7]. The post-irradiation fracture toughness properties of the reactor vessel belt line material were obtained directly from the Kewaunee Reactor Vessel Material Surveillance Program.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section ~~III~~ XI of the ASME Boiler and Pressure Vessel Code, and are discussed in detail in Footnote.[8]

The method specifies that the allowable total stress intensity factor (K_I) at any time during heatup or cooldown cannot be greater than that shown on the $K_{I\cancel{R}c}$ curve for the metal temperature at that time. Furthermore, the approach applies an explicit safety factor of 2.0 on the stress intensity factor induced by the pressure gradient. Thus, the governing equation for the heatup-cooldown analysis is:

$$2\,K_{Im} + K_{It} \leq K_{I\cancel{R}c} \qquad (3.1b\text{-}1)$$

where

K_{Im} is the stress intensity factor caused by membrane (pressure) stress

K_{It} is the stress intensity factor caused by the thermal gradients

$K_{I\cancel{R}c}$ is provided by the Code as a function of temperature relative to the RT_{NDT} of the material.

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[5] Section III and XI of the ASME Boiler and Pressure Vessel Code, Appendix G, "Protection Against Non-ductile Failure."

[6] Standard Method for Measuring Thermal Neutron Flux by Radioactivation Techniques, ASTM designation E262-86.

[7] Exemption From the Requirements of 10 CFR 50, Appendix G, Appendix H, and Section 50.61, dated May 1, 2001.

[8] WCAP-~~14278, Revision 1, "Kewaunee Heatup and Cooldown Limit Curves for Normal Operation," T. Laubham and C. Kim, September 1998~~16643-NP, Revision 2, "Kewaunee Power Station Heatup and Cooldown Limit Curves for Normal Operation", dated June 2009.

From equation (3.1b-1) the variables that affect the heatup and cooldown analysis can be readily identified. K_{Im} is the stress intensity factor due to membrane (pressure) stress. K_{It} is the thermal (bending) stress intensity factor and accounts for the linearly varying stress in the vessel wall due to thermal gradients. During heatup K_{It} is negative on the inside and positive on the outer surface of the vessel wall. The signs are reversed for cooldown and, therefore, an ID or an OD one quarter thickness surface flaw is postulated in whichever location is more limiting. K_{IRc} is dependent on irradiation and temperature and, therefore, the fluence profile through the reactor vessel wall and the rates of heatup and cooldown are important. The heatup and cooldown limit curves have been developed by combining the most conservative pressure temperature limits derived by using material properties of the ~~intermediate~~ upper shell forging, ~~closure head~~vessel flange, and beltline circumferential weld to form a single set of composite curves. Details of the procedure used to account for these variables are explained in the following text.

Following the generation of pressure-temperature curves for both the steady-state (zero rate of change of temperature) and finite heatup rate situations, the final limit curves are produced in the following fashion. First, a composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data for each of the limiting materials. At any given temperature, the allowable pressure is taken to be the lesser of the values taken from the curves under consideration. The composite curve is then adjusted to allow for possible errors in the pressure and temperature sensing instruments including the pressure difference between the gage and beltline weld.

The use of the composite curve is mandatory in setting heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling analysis switches from the OD to the ID location. The pressure limit must, at all times, be based on the most conservative case.

The cooldown analysis proceeds in the same fashion as that for heatup with the exception that the controlling location is always at the ID. The thermal gradients induced during cooldown tend to produce tensile stresses at the ID location and compressive stresses at the OD position. Thus, the ID flaw is clearly the worst case.

As in the case of heatup, allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations for each of the limiting materials. Composite limit curves are then constructed for each cooldown rate of interest. Again, adjustments are made to account for pressure and temperature instrumentation error.

The use of the composite curve in the cooldown analysis is necessary because system control is based on a measurement of reactor coolant temperature, whereas the limiting pressure is calculated using the material temperature at the tip of the assumed reference flaw. During cooldown, the 1/4T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition, of course, is not true for the steady-state situation. It follows that the ΔT induced during cooldown results in a calculated higher K_{IRc} for finite cooldown rates than for steady-state under certain conditions.

Limit curves for normal heatup and cooldown of the primary Reactor Coolant System have been calculated using the methods discussed above and limited application to ASME Boiler and Pressure Vessel Code Case N-588 to the circumferential beltline weld. The derivation of the limit curves is consistent with the NRC Regulatory Standard Review Plan[9] and Footnote.[10]

Transition temperature shifts occurring in the pressure vessel materials due to radiation exposure have been obtained directly from the reactor pressure vessel surveillance program. As presented in WCAP ~~14279~~16641-NP, Revision ~~1~~0[11], weld metal Charpy test specimens from Capsule ~~S~~ T indicate that the core region weld metal exhibits the largest shift in RT_{NDT} (~~250~~271°F).

The results of Irradiation Capsules V, R, P, ~~and~~ S, and T analyses are presented in WCAP 8908,[12] WCAP 9878,[13] WCAP-12020,[14] WCAP-14279,[15] ~~and~~ WCAP-14279, Revision 1[16], and WCAP 16641-NP respectively. Heatup and cooldown limit curves for normal operation of the reactor vessel are presented in Figures TS 3.1-1 and TS 3.1-2 and represent an operational time period of ~~33~~52.1~~[1]~~ effective full power years.

The isothermal cooldown limit curve (Figure TS 3.1-2) and 5°F/hr heatup limit curve (Figure TS 3.1-1) are ~~is~~ used for evaluation of low temperature overpressure protection (LTOP) events. ~~This~~ These curves ~~is~~ are applicable for ~~33[1]~~52.1 effective full-power years of fluence (~~through the end of OPERATING cycle 33[1]~~projected to coincide with the beginning of fuel cycle 46 based on continuous 18 month fuel cycles). If a low temperature overpressure event occurred, the RCS pressure transient would be evaluated to the limits of this figure to verify the integrity of the reactor vessel. If these limits are not exceeded, vessel integrity is assured and a TS violation has not occurred.

~~Note:~~

~~[1] The curves are limited to 31.1 EFPY due to changes in vessel fluence associated with operation at uprated power.~~

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(9) "Fracture Toughness Requirements", Branch Technical Position MTEB 5-2, Chapter 5.3.2 in Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition, NUREG-0800, 1981.

(10) 1989 ASME Boiler and Pressure Vessel (B&PV) Code, Section XI, Appendix G, "Fracture Toughness Criteria for Protection Against Failure."

(11) B. N. Burgos, "Analysis of Capsule T from Dominion Energy Kewaunee Power Station Reactor Vessel Radiation Surveillance Program", WCAP-16641-NP, Revision 0, October 2006

(12) S.E. Yanichko, S. L. Anderson, and K. V. Scott, "Analysis of Capsule V from the Wisconsin Public Service Corporation Kewaunee Nuclear Plant Reactor Vessel Radiation Surveillance Program," WCAP 8908, January 1977.

(13) S.E. Yanichko, et al., "Analysis of Capsule R from the Wisconsin Public Service Corporation Kewaunee Nuclear Plant Reactor Vessel Radiation Surveillance Program," WCAP 9878, March 1981.

(14) S.E. Yanichko, et al., "Analysis of Capsule P from the Wisconsin Public Service Corporation Kewaunee Nuclear Power Plant Reactor Vessel Radiation Surveillance Program," WCAP-12020, November 1988.

(15) E. Terek, et al., "Analysis of Capsule S from the Wisconsin Public Service Corporation Kewaunee Nuclear Power Plant Reactor Vessel Radiation Surveillance Program," WCAP-14279, March 1995.

(16) C. Kim, et al., "Evaluation of Capsule S from the Kewaunee and Capsule A35 from the Maine Yankee Nuclear Power Reactor Vessel Radiation Surveillance Programs," WCAP-14279, Revision 1, September 1998.

Pressurizer Limits (TS 3.1.b.3)

Although the pressurizer operates at temperature ranges above those for which there is reason for concern about brittle fracture, OPERATING limits are provided to ensure compatibility of operation with the fatigue analysis performed in accordance with Code requirements. In-plant testing and calculations have shown that a pressurizer heatup rate of 100°F/hr cannot be achieved with the installed equipment.

Low Temperature Overpressure Protection (TS 3.1.b.4)

The Low Temperature Overpressure Protection System must be OPERABLE during startup and shutdown conditions below the enable temperature (i.e., low temperature) as defined in Branch Technical Position RSB 5-2 as modified by ASME Boiler and Pressure Vessel Code ~~Case N-514~~Appendix G. Based on the Kewaunee Appendix G LTOP protection pressure-temperature limits calculated through ~~33[1]~~52.1 effective full-power years, the LTOP System must be OPERABLE whenever one or more of the RCS cold leg temperatures are ≤ ~~200~~343°F (356°F indicated to account for instrument uncertainty) and the head is on the reactor vessel. The LTOP system is considered OPERABLE when all four valves on the RHR suction piping (valves RHR-1A, 1B, 2A, 2B) are open and valve RHR-33-1, the LTOP valve, is able to relieve RCS overpressure events without violating Figure TS 3.1-2.

The set pressure specified in TS 3.1.b.4 includes consideration for the opening pressure tolerance of ± 3% (± 15 psig) as defined in ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NC: Class 2 Components for Safety Relief Valves. The analysis of pressure transient conditions has demonstrated acceptable relieving capability at the upper tolerance limit of 515 psig.

If one train of RHR suction piping to RHR 33-1 is isolated, then the valves and valve breakers in the other train shall be verified open, and the isolated flowpath must be restored within five days. If the isolated flowpath cannot be restored within five days, then the RCS must be depressurized and vented through at least a 6.4 square inch vent within an additional eight hours.

If both trains of RHR suction are isolated or valve RHR 33-1 is inoperable, then the system can still be considered OPERABLE if an alternate vent path is provided which has the same or greater effective flow cross section as the LTOP safety valve (≥ 6.4 square inches). If vent path is provided by physical openings in the RCS pressure boundary (e.g., removal of pressurizer safety valves or steam generator manways), then the vent path is considered secured in the open position.

~~Note~~

~~[1] The curves are limited to 31.1 EFPY due to changes in vessel fluence associated with operation at uprated power.~~